June 17, 1982

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Docket No. 50-277

Mr. Edward G. Bauer, Jr.
Vice President and General Counsel
Philadelphia Electric Company
2301 Market Street
Philadelphia, Pennsylvania 19101

Dear Mr. Bauer:

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-44 for the Peach Bottom Atomic Power Station, Unit No. 2. The amendment revises the Technical Specifications (TSs) in response to your application dated February 19, 1982, as amended by your letter dated June 3, 1982.

The changes to the TSs permit reactor operation of Peach Bottom Unit No. 2 with the Reload Number 5 core (Cycle 6).

Copies of our Safety Evaluation and a related Notice of Issuance are also enclosed.

Sincerely,

Morton B. Fairtile, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 86 to DPR-44
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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MBF

No legal objection to form of notice or amendment. SER review not requested.

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD:OR:DL	OELD		
SURNAME	RIngram	MFairtile;cf	JStolz	TNovak	CUTCHIN		
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
June 17, 1982

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Docket No. 50-277

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

XX Notice of Issuance of Facility Operating License(s) or Amendment(s).

XX Other: Amendment No. 86.
Referenced documents have been provided PDR.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE →	ORB#4:DL					
SURNAME →	RIngram;cf					
DATE →	6/17/82					

NRC FORM 102 7-79

Philadelphia Electric Company

cc w/enclosure(s):

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cc w/enclosure(s) & incoming dtd.:
2/19/82, 6/3/82

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-277

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-44

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Philadelphia Electric Company, et al. (the licensee) dated February 19, 1982, as supplemented June 3, 1982, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-44 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86, are hereby incorporated in the license. PECO shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 17, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-44

DOCKET NO. 50-277

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove Pages	Insert Pages
iv	iv
—	iva
vi	vi
9	9
13	13
14	14
103	103
104	104
133a	133a
133b	133b
133c	133c
—	133d
—	133e
140	140
140a	140a
140b	140b
140c	140c
140d	140d
140e	—

Remove Pages	Insert Pages
142	142
142a	142a
142b	142b
142c	—
142f	142f
142g	142g
142h	142h
—	142i
—	142j

LIST OF FIGURES

LIST OF FIGURES

LIST OF TABLES

Amendment No. ~~33~~, ~~48~~,86

SAFETY LIMIT

1.1 FUEL CLADDING INTEGRITY

Applicability:

The Safety Limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objectives:

The objective of the Safety Limits is to establish limits which assure the integrity of the fuel cladding.

Specification:

A. Reactor pressure ≥800 psia and Core Flow ≥10% of Rated

The existence of a minimum critical power ratio MCPR less than 1.07 for two recirculation loop operation, or 1.08 for single loop operation, shall constitute violation of the fuel cladding integrity safety limit.

To ensure that this safety limit is not exceeded, neutron flux shall not be above the scram setting established in specification 2.1.A for longer than 1.15 seconds as indicated by the process computer. When the process computer is out of service this safety limit shall be assumed to be exceeded if the neutron flux exceeds its scram setting and a control rod scram does not occur.

LIMITING SAFETY SYSTEM SETTING

2.1 FUEL CLADDING INTEGRITY

Applicability:

The Limiting Safety System Settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity Safety Limits from being exceeded.

Objectives:

The objective of the Limiting Safety System Settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity Safety Limits from being exceeded.

Specification:

The limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM Flux Scram Trip Setting (Run Mode)

When the Mode Switch is in the RUN position, the APRM flux scram trip setting shall be:

$S \leq 0.66W + 54\% - 0.66\Delta W$

where:

S = Setting in percent of rated thermal power (3293 MWt)

W = Loop recirculating flow rate in percent of design. W is 100 for core flow of 102.5 million lb/hr or greater.

Amendment No. 13, 34, 42, 48, 78, 86

1.1 BASES: FUEL CLADDING INTEGRITY

A. Fuel Cladding Integrity Limit at Reactor Pressure ≥ 800 psia and Core Flow ≥ 10% of Rated

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the critical power ratio in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis described in references 1 and 3.

1.1.A BASES (Cont'd)

B. Core Thermal Power Limit (Reactor Pressure < 800 psia on Core Flow < 10% of Rated)

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is established by other means. This is done by establishing a limiting condition of core thermal power operation with the following basis.

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28 x 10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this bundle power corresponds to a core thermal power of more than 50%. Therefore, a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10 % is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 1.1.A or 1.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage.

LIMITING CONDITIONS FOR OPERATION

3.3.B Control Rods (Cont'd)

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per second.

5. During operation with limiting control rod patterns as determined by the designated qualified personnel, either:

   a. Both RMB channels shall be operable, or
   b. Control rod withdrawal shall be blocked, or
   c. The operating power level shall be limited so that the MCPR will remain above the fuel cladding integrity safety limit assuming a single error that results in complete withdrawal of a single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted from Fully Withdrawn	Avg. Scram Insertion Times (sec)
5	0.375
20	0.90
50	2.0
90	3.5

SURVEILLANCE REQUIREMENTS

4.3.B Control Rods (Cont'd)

4. Prior to control rod withdrawal for startup or during refueling verify that at least two sources range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s).

C. Scram Insertion Times

1. After each refueling outage, and prior to synchronizing the main turbine generator initially following restart of the plant, all operable fully withdrawn insequence rods shall be scram time tested during startup from the fully withdrawn position with the nuclear system pressure above 800 psig.

Amendment No. 73, 48, 86

LIMITING CONDITIONS FOR OPERATION

3.3.C (Cont'd)

2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two-by-two array shall be no greater than:

% Inserted From Fully Withdrawn	Avg. Scram Insertion Times (Sec)
5	0.398
20	0.954
50	2.120
90	3.8

3. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds.

SURVEILLANCE REQUIREMENTS

4.3.C (Cont'd)

After exceeding 30 percent power all previously untested operable control rods shall be tested as described above prior to exceeding 40 percent power.

2. Whenever such scram time measurements are made (such as when a scram occurs and the scram insertion time recorders are operable) an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

Amendment No. 86

LIMITING CONDITIONS FOR OPERATION

3.5.I Average Planar LHGR

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in the applicable figures during two recirculation loop operations. During single loop operation, the APLHGR for each fuel type shall not exceed the above values multiplied by the following reduction factors: 0.71 for 7x7 fuel; 0.83 for 8x8 fuel; 0.81 for LTA, 8X8R and P8X8R fuel. If at any time during operation it is determined by normal surveillance that the limiting value of APLHGR is being exceeded, action shall be initiated within one (1) hour to restore APLHGR to within prescribed limits. If the APLHGR is not returned to within prescribed limits within five (5) hours reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless APLHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

3.5.J Local LHGR

During power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed design LHGR.

$LHGR \leq LHGRd$

$LHGR =$ Design LHGR

13.4 kW/ft for all 8x8 fuel

SURVEILLANCE REQUIREMENTS

4.5.I Average Planar LHGR

The APLHGR for each type of fuel as a function of average planar exposure shall be checked daily during reactor operation at $\geq$25% rated thermal power.

4.5.J Local LHGR

The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$25% rated thermal power.

Amendment No. ~~40~~, ~~48~~, ~~70~~, ~~78~~, 86

LIMITING CONDITIONS FOR OPERATION

3.5.J Local LHGR (Cont'd)
If at any time during operation it is determined by normal surveillance that limiting value for LHGR is being exceeded, action shall be initiated within one (1) hour to restore LHGR to within prescribed limits. If the LHGR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless LHGR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the precribed limits.

3.5.K. Minimum Critical Power Ratio (MCPR)

1. During power operation the MCPR for the applicable incremental cycle core average exposure and for each type of fuel shall be equal to or greater than the value given in Specification 3.5.K.2 or 3.5.K.3 times Kf, where Kf is as shown in Figure 3.5.1.E. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within one (1) hour to restore MCPR to within prescribed limits. If the MCPR is not returned to within prescribed limits within five (5) hours, reactor power shall be decreased at a rate which would bring the reactor to the cold shutdown condition within 36 hours unless MCPR is returned to within limits during this period. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

SURVEILLANCE REQUIREMENTS

4.5.K Minimum Critical Power Ratio (MCPR)

1. MCPR shall be checked daily during reactor power operation at $\geq 25\%$ rated thermal power.
2. Except as provided in Specification 3.5.K.3, the verification of the applicability of 3.5.K.2.a Operating Limit MCPR Values shall be performed every 120 operating days by scram time testing 19 or more control rods on a rotation basis and performing the following:

a. The average scram time to the 20% insertion position shall be:

$$\tau_{ave} \leq \tau_B$$

b. The average scram time to the 20% insertion position is determined as follows:

$$\tau_{ave} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

where: n = number of surveillance tests performed to date in the cycle

Amendment No. ~~38~~, ~~48~~, 86

LIMITING CONDITIONS FOR OPERATION

3.5.K. Minimum Critical Power Ratio (MCPR) (Cont'd)

2. Except as specified in 3.5.K.3, the Operating Limit MCPR Values are as follows:

   a. If requirement 4.5.K.2.a is met:
   The Operating Limit MCPR values are as given in Table 3.5.K.2.

   b. If requirement 4.5.K.2.a is not met:
   The Operating Limit MCPR values as a function of τ are as given in Figures 3.5.K.1, 3.5.K.2, and 3.5.K.3.

Where:

$$\tau = \frac{\tau ave - \tau B}{0.90 - \tau B}$$

3. The Operating Limit MCPR values shall be as given in Table 3.5.K.3 if the Surveillance Requirement of Section 4.5.K.2 to scram time test control rods is not performed.

SURVEILLANCE REQUIREMENTS

4.5.K. Minimum Critical Power Ratio (MCPR) (Cont'd)

Ni = number of active control rods measured in the ith surveillance test.

τi = average scram time to the 20% insertion position of all rods measured in the ith surveillance test.

c. The adjusted analysis mean scram time (τB) is calculated as follows:

$$\tau B = \mu + 1.65 \left(\frac{Ni}{\sum_{i=1}^{n} Ni} \right)^{1/2} \sigma$$

Where:

μ = mean of the distribution for average scram insertion time t the 20% position - 0.710 sec.

Ni = total number of active control rods measured in specification 4.3.C.1

σ = standard deviation of the distribution for average scram insertion time to the 20% position = 0.053.

Amendment No. 86

Table 3.5.K.2

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

Fuel Type	MCPR Operating Limit** For Incremental Cycle Core Average Exposure	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
8x8R/LTA	1.23	1.27
P 8x8R	1.25	1.30
P8DRB285	1.29	1.30

* If requirement 4.5.K.2.a is met.

** These values shall be increased by 0.01 for single loop operation.

Table 3.5.K.3

OPERATING LIMIT MCPR VALUES
FOR VARIOUS CORE EXPOSURES*

Fuel Type	MCPR Operating Limit** For Incremental Cycle Core Average Exposure	
	BOC to 2000 MWD/t Before EOC	2000 MWD/t before EOC To EOC
8x8R/LTA	1.34	1.39
P 8x8R	1.37	1.42
P8DRB285	1.37	1.42

* If surveillance requirement 4.5.K.2 is not performed.

** These values shall be increased by 0.01 for single loop operation.

3.5 BASES (Cont'd.)

H. Engineering Safeguards Compartments Cooling and Ventilation

One unit cooler in each pump compartment is capable of providing adequate ventilation flow and cooling. Engineering analyses indicated that the temperature rise in safeguards compartments without adequate ventilation flow or cooling is such that continued operation of the safeguards equipment or associated auxiliary equipment cannot be assured. Ventilation associated with the High Pressure Service Water Pumps is also associated with the Emergency Service Water pumps, and is specified in Specification 3.9.

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR Part 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is the LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in the applicable figure for each fuel type.

The calculational procedure used to establish the APLHGR is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (G.E.) calculational models which are consistent with the requirements of Appendix K to 10CFR Part 50. A complete discussion of each code employed in the analysis is presented in Reference 4. Input and model changes in the Peach Bottom loss-of-coolant analysis which are different from the previous analyses performed with Reference 4 are described in detail in Reference 8. These changes to the analysis include: (1) consideration of the counter current flow limiting (CCFL) effect, (2) corrected code inputs, and (3) the effect of drilling alternate flow paths in the bundle lower tie plate.

Amendment No. ~~23~~, ~~30~~, ~~40~~, ~~48~~, ~~70~~, 86

3.5.I BASES (Cont'd)

J. Local LHGR

This specification assures that the linear heat generation rate in any 8X8 fuel rod is less than the design linear heat generation. The maximum LHGR shall be checked daily during reactor operation at ≥25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be at the design LHGR below 25% rated thermal power, the peak local LHGR must be a factor of approximately ten (10) greater than the average LHGR which is precluded by a considerable margin when employing any permissible control rod pattern.

K. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions are derived from the established fuel cladding integrity Safety Limit MCPR and analyses of the abnormal operational transients presented in Supplemental Reload Licensing Analysis and Reference 7. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.1

To assure that the fuel cladding integrity Safety Limit is not violated during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The transients evaluated are as described in reference 7.

3.5.K. BASES (Cont'd)

The largest reduction in critical power ratio is then added to the fuel cladding integrity safety limit MCPR to establish the MCPR Operating Limit for each fuel type.

Two codes are used to analyze the rod withdrawal error transient. The first code simulates the three dimensional BWR core nuclear and thermal-hydraulic characteristics. Using this code a limiting control rod pattern is determined; the following assumptions are included in this determination:

(1) The core is operating at full power in the xenon-free condition.

(2) The highest worth control rod is assumed to be fully inserted.

(3) The analysis is performed for the most reactive point in the cycle.

(4) The control rods are assumed to be the worst possible pattern without exceeding thermal limits.

(5) A bundle in the vicinity of the highest worth control rod is assumed to be operating at the maximum allowable linear heat generation rate.

(6) A bundle in the vicinity of the highest worth control rod is assumed to be operating at the minimum allowable critical power ratio.

The three-dimensional BWR code then simulates the core response to the control rod withdrawal error. The second code calculates the Rod Block Monitor response to the rod withdrawal error. This code simulates the Rod Block Monitor under selected failure conditions (LPRM) for the core response (calculated by the 3-dimensional BWR simulation code) for the control rod withdrawal.

The analysis of the rod withdrawal error for Peach Bottom Unit 2 considers the continuous withdrawal of the maximum worth control rod at its maximum drive speed from the reactor which is operating with the limiting control rod pattern as discussed above.

3.5.K. BASES (Cont'd)

A brief summary of the analytical method used to determine the nuclear characteristics is given in Section 3 of Reference 7.

Analysis of the abnormal operational transients is presented in Section 5.2 of Reference 7. Input data and operating conditions used in this analysis are shown in Table 5-8 of Reference 7 and in the Supplemental Reload Licensing Analysis.

L. Average Planar LHGR (APLHGR), Local LHGR and Minimum Critical Power Ratio (MCPR)

In the event that the calculated value of APLHGR, LHGR or MCPR exceeds its limiting value, a determination is made to ascertain the cause and initiate corrective action to restore the value to within prescribed limits. The status of all indicated limiting fuel bundles is reviewed as well as input data associated with the limiting values such as power distribution, instrumentation data (Traversing In-Core Probe TIP, Local Power Range Monitor - LPRM, and reactor heat balance instrumentation), control rod configuration, etc., in order to determine whether the calculated values are valid.

In the event that the review indicates that the calculated value exceeding limits is valid, corrective action is immediately undertaken to restore the value to within prescribed limits. Following corrective action, which may involve alternations to the control rod configuration and consequently changes to the core power distribution, revised instrumentation data, including changes to the relative neutron flux distribution,for up to 43 incore locations is obtained and the power distribution, APLHGR, LHGR and MCPR calculated. Corrective action is initiated within one hour of an indicated value exceeding limits and verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication.

In the event that the calculated value of APLHGR, LHGR or MCPR exceeding its limiting value is not valid, i.e., due to an erroneous instrumentation indication, etc., corrective action is initiated within one hour of an indicated value exceeding limits. Verification that the indicated value is within prescribed limits is obtained within five hours of the initial indication. Such an invalid indication would not be a violation of the limiting condition for operation and therefore would not constitute a reportable occurrence.

3.5.L BASES (Cont'd)

Operating experience has demonstrated that a calculated value of APLHGR, LHGR or MCPR exceeding its limiting value predominately occurs due to this latter cause. This experience coupled with the extremely unlikely occurrence of concurrent operation exceeding APLHGR, LHGR or MCPR and a Loss of Coolant Accident or applicable Abnormal Operational Transients demonstrates that the times required to initiate corrective action (1 hour) and restore the calculated value of APLHGR, LHGR or MCPR to within prescribed limits (5 hours) are adequate.

3.5.M. References

1. "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel", Supplements 6,7, and 8 NEDM-10735, August 1973.

2. Supplement 1 to Technical Report on Densifications of General Electric Reactor Fuels, December 14, 1974 (Regulatory Staff).

3. Communication: V. A. Moore to I. S. Mitchell, "Modified GE Model for Fuel Densification", Docket 50-321, March 27, 1974.

4. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE 20566 (Draft), August 1974.

5. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to the USAEC by latter, G. L. Gyorey to Victor Stello, Jr., dated December 20, 1974.

6. DELETED

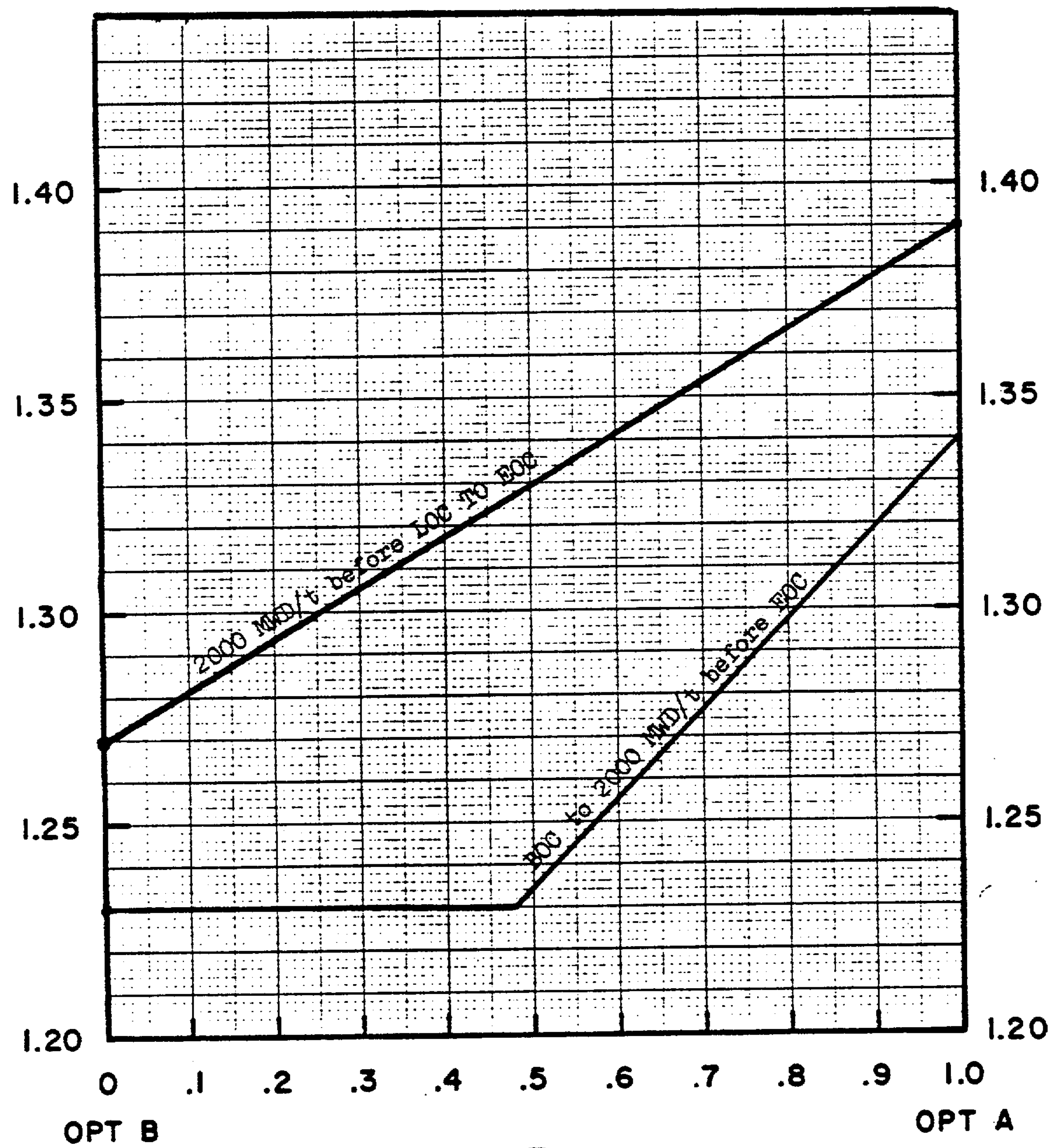
7. General Electric Boiling Water Reactor Generic Reload Fuel Application. NEDO-24011-P-A.

8. Loss-of-Coolant Accident Analysis for Peach Bottom Atomic Power Station Unit 2, NEDO-24081, December 1977, and for Unit 3, NEDO-24082, December, 1977.

PEACH BOTTOM UNIT 2

FIGURE 3.5.K.1 MCPR OPERATING LIMIT vs τ

FUEL TYPE 8x8R/LTA



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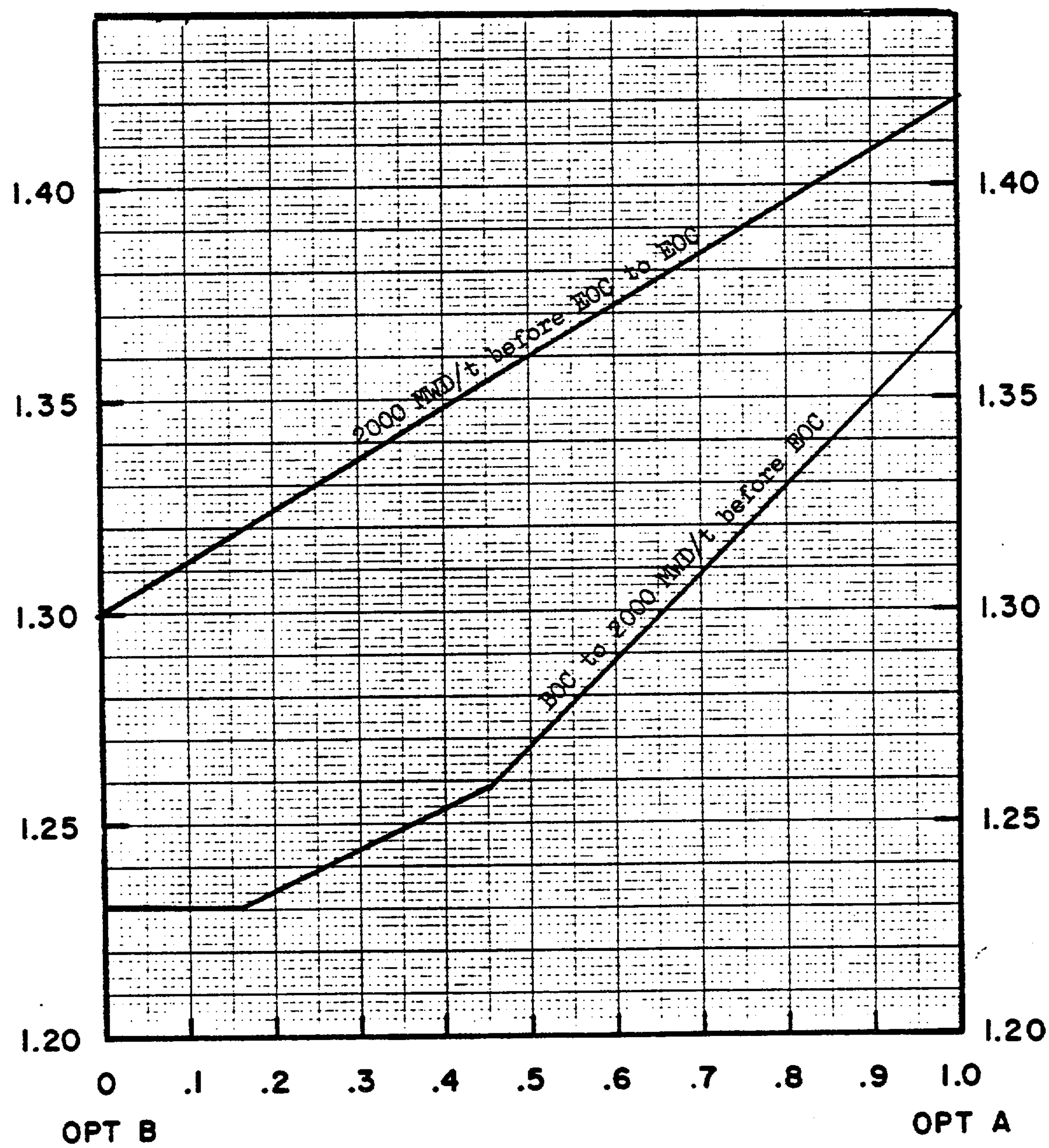
PEACH BOTTOM UNIT 2

FIGURE 3.5.K.2 MCPR OPERATING LIMIT vs τ

FUEL TYPE P8x8R



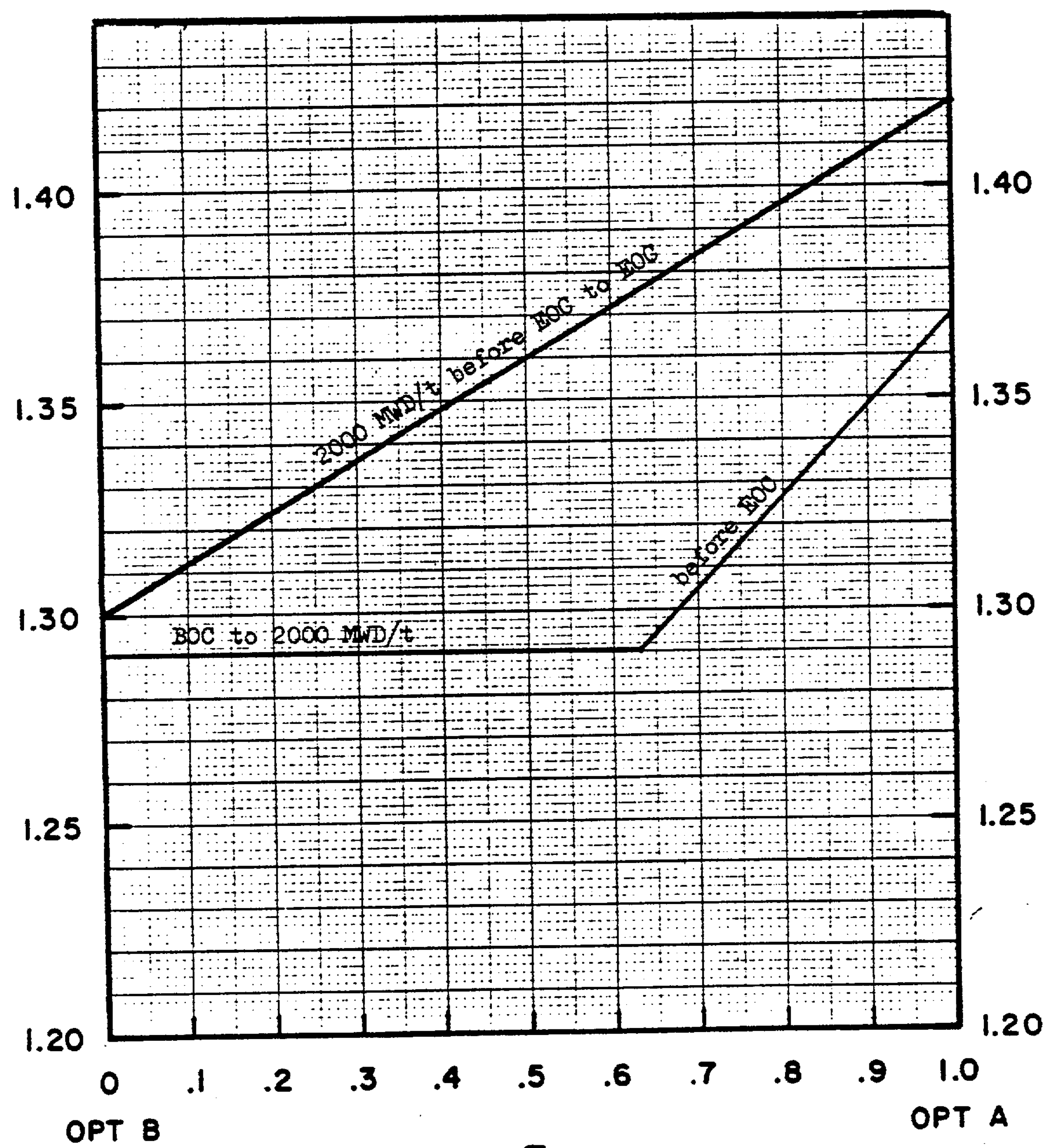
τ

Amendment No. ~~16~~, ~~40~~, ~~70~~, 86

PEACH BOTTOM UNIT 2

FIGURE 3.5.K.3 MCPR OPERATING LIMIT vs τ

FUEL TYPE P8DRB285



Amendment No. ~~16~~, ~~40~~, ~~70~~, ~~79~~, 86

PEACH BOTTOM UNIT 2

8X8R FUEL

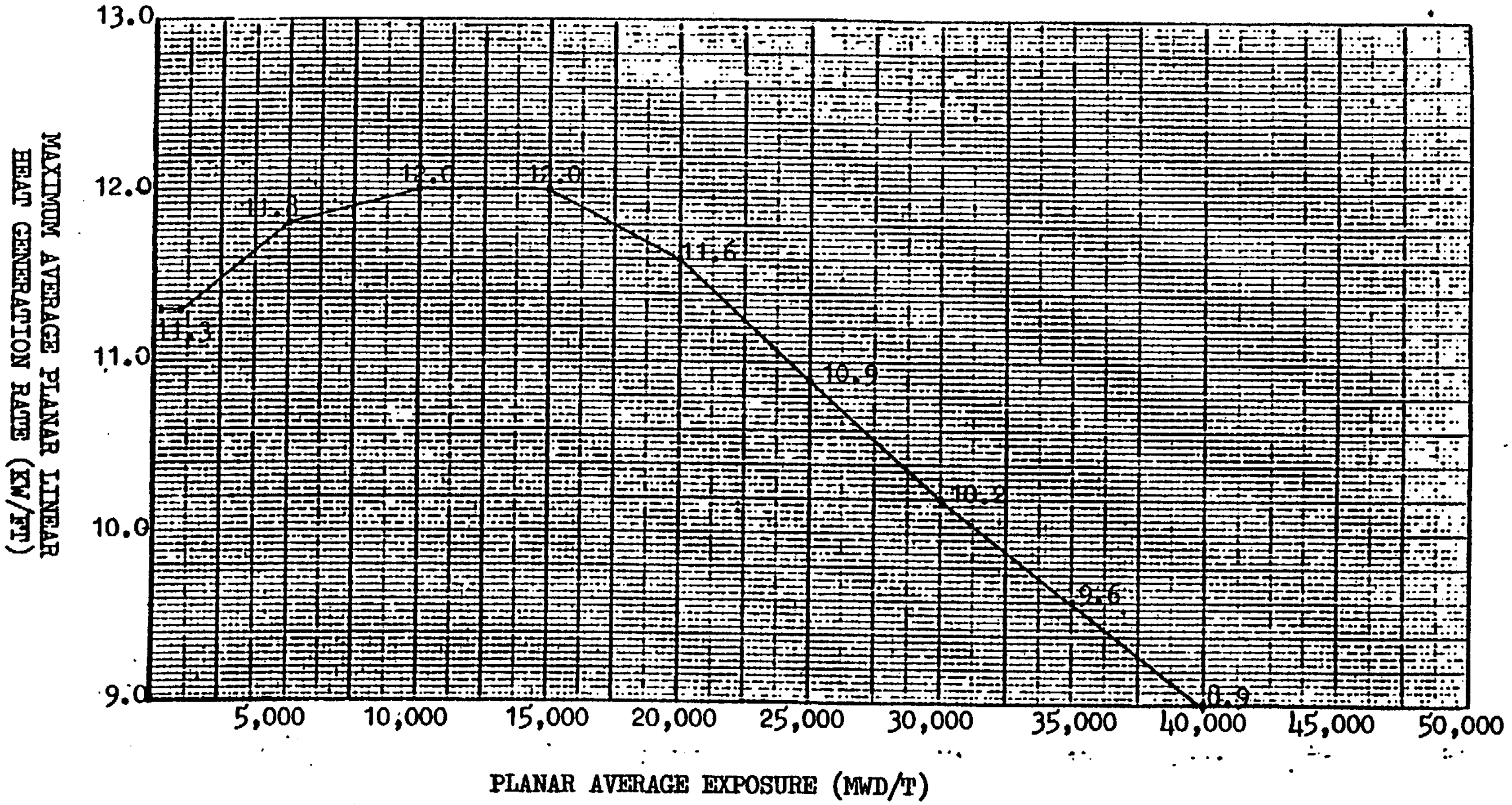


FIGURE 3.5.1.G Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

P8X8R FUEL
TYPE P8DRB285 (Applicable to 100 mil Channels)

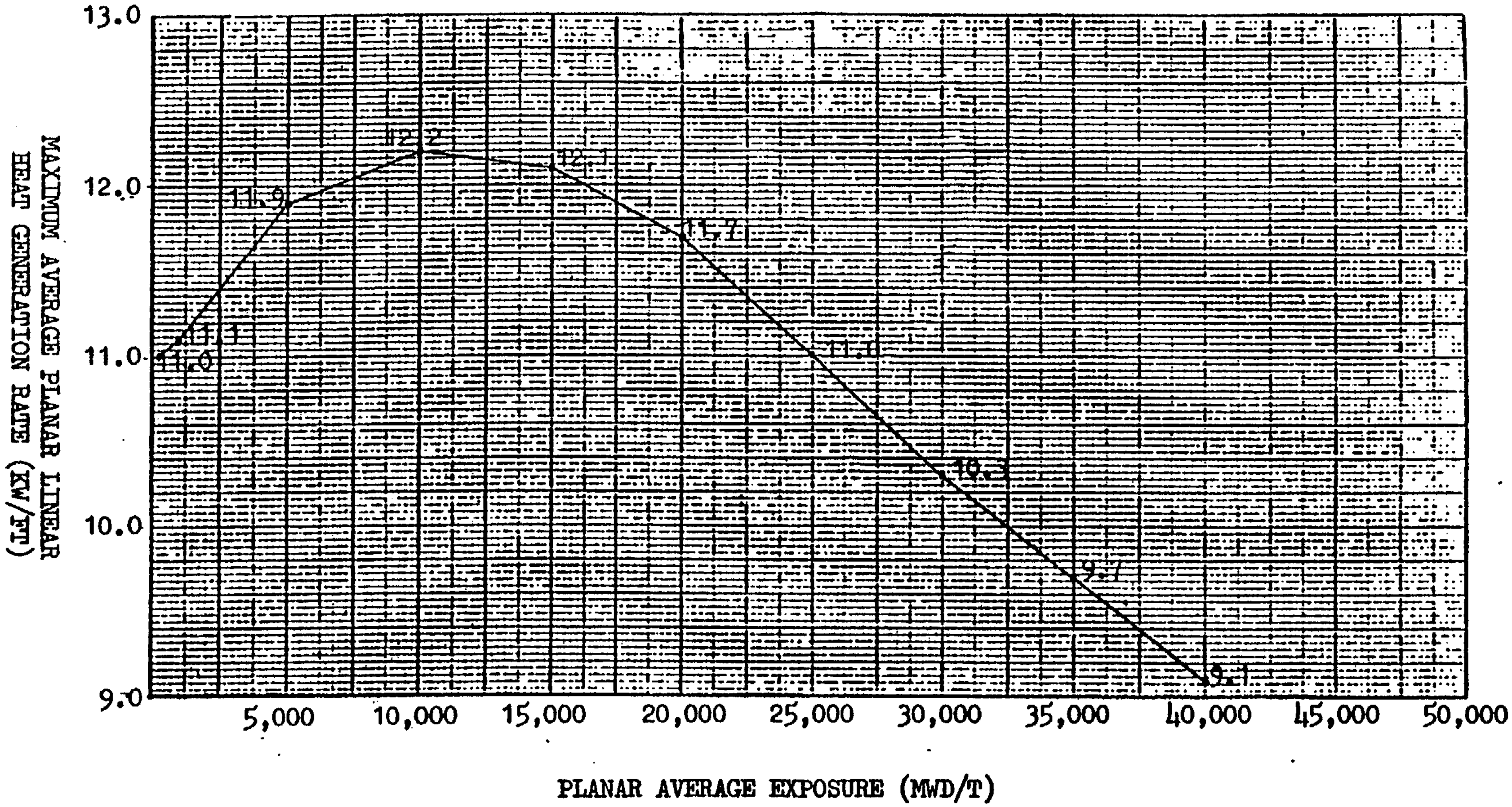
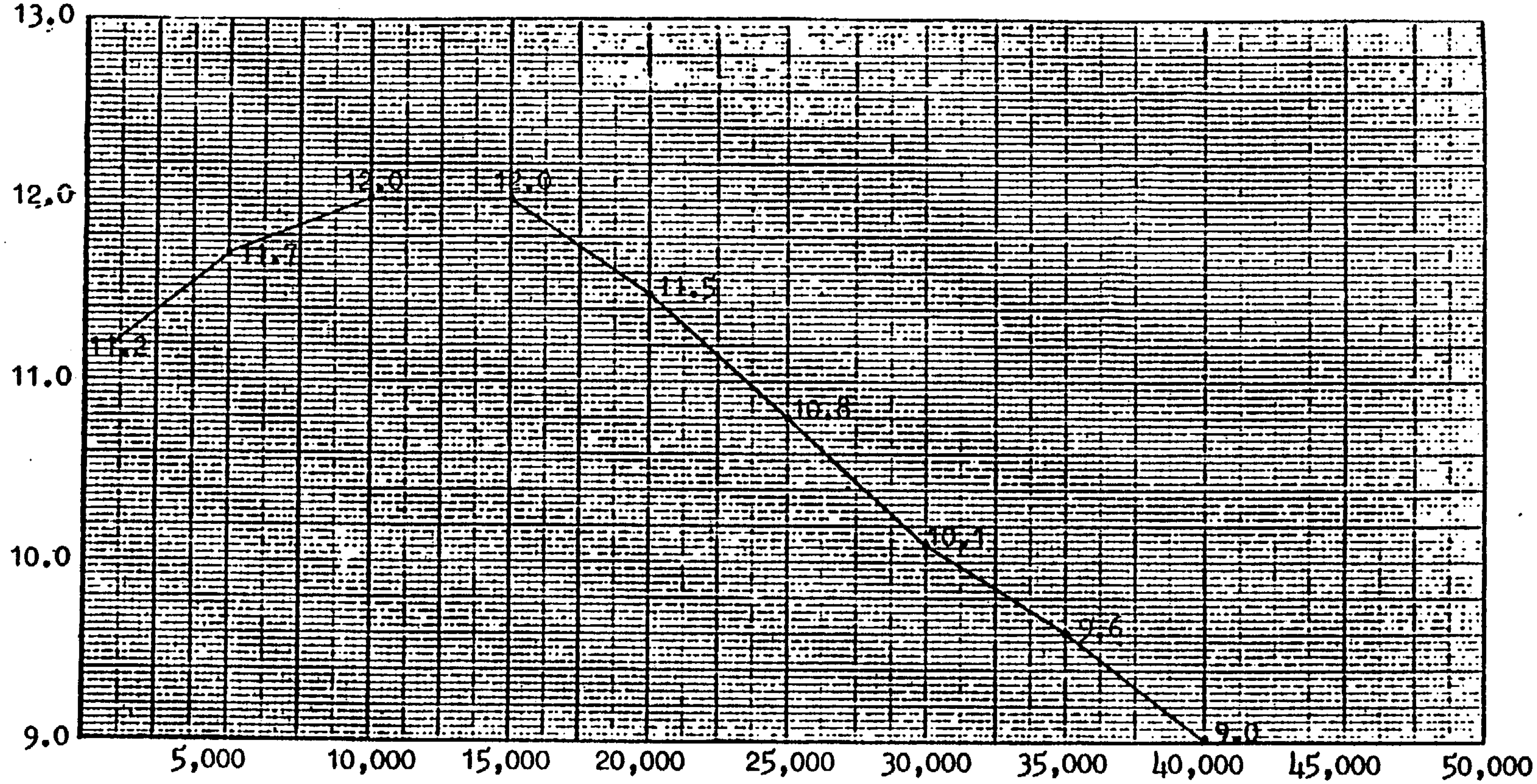


FIGURE 3.5.1.H Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

P8X8R FUEL
TYPE P8DRB284H
(Applicable to 80 mil, 100 mil, and 120 mil Channels)

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (KW/FT)



PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.I Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

P8X8R FUEL TYPE P8DRB299

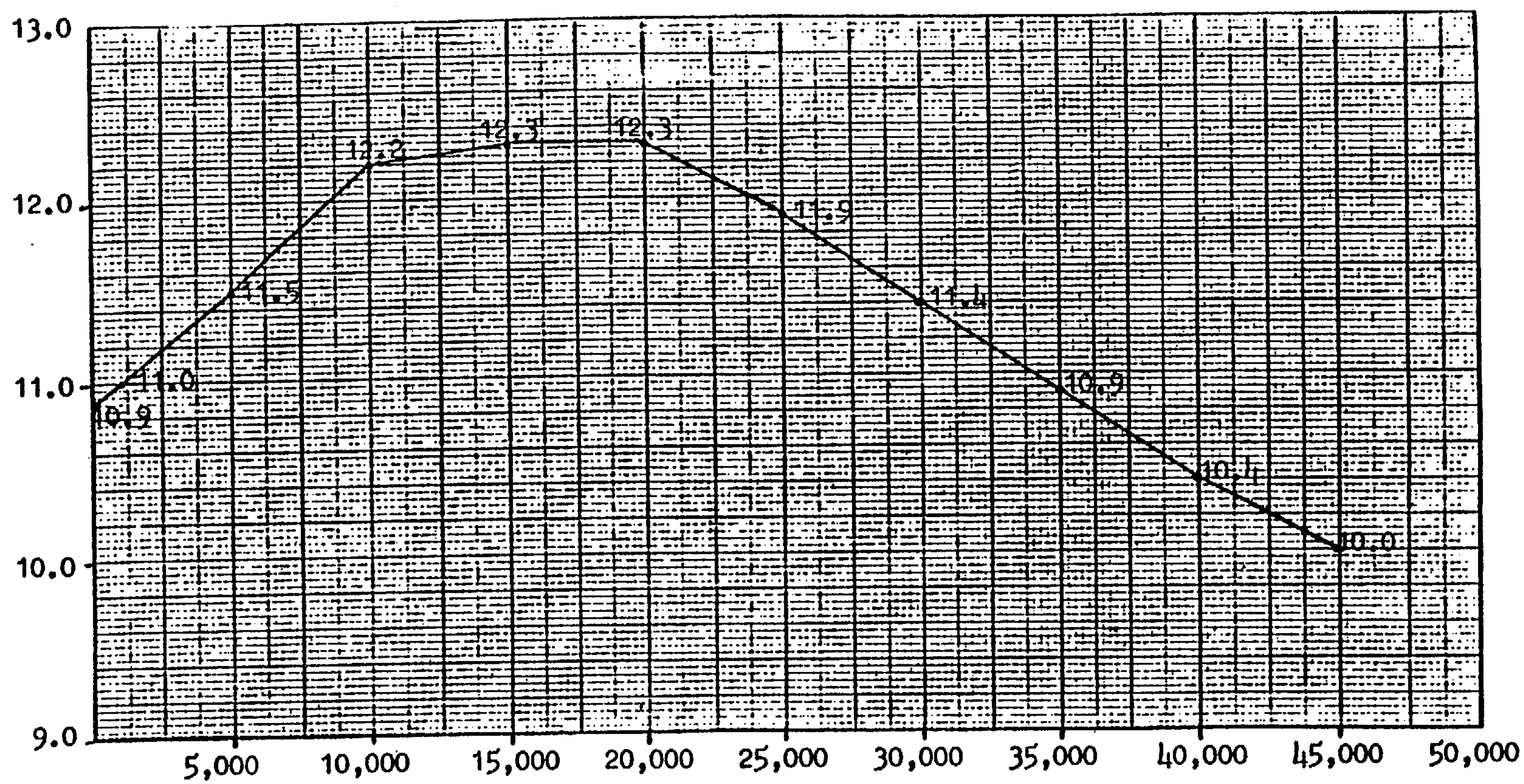


FIGURE 3.5.1.J Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

PEACH BOTTOM UNIT 2

P8X8R GENERIC

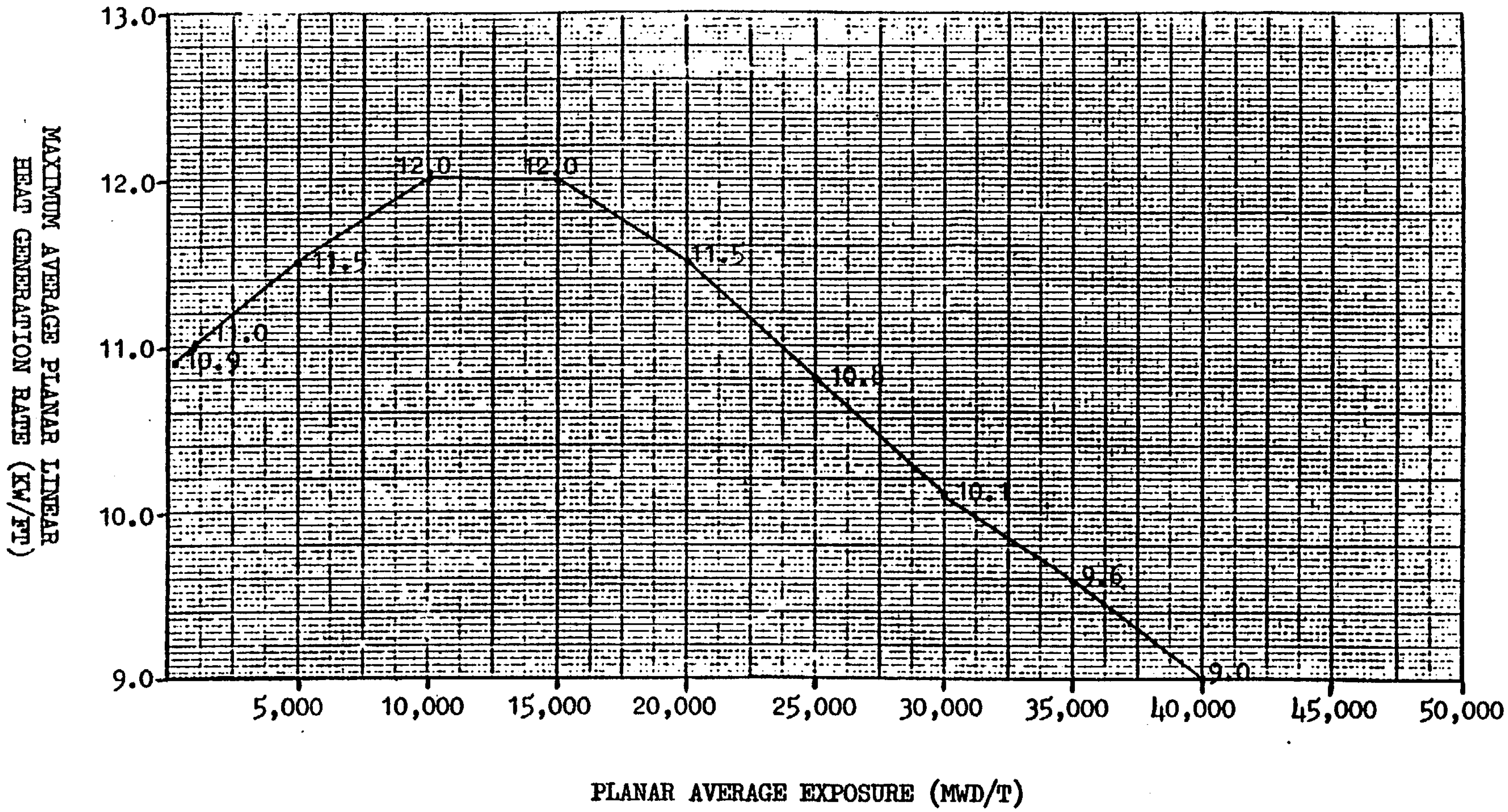


FIGURE 3.5.1.K Maximum Average Planar Linear Heat Generation Rate Versus Planar Average Exposure

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-44

PHILADELPHIA ELECTRIC COMPANY
PUBLIC SERVICE ELECTRIC AND GAS COMPANY
DELMARVA POWER AND LIGHT COMPANY
ATLANTIC CITY ELECTRIC COMPANY

PEACH BOTTOM ATOMIC POWER STATION, UNIT NO. 2

DOCKET NO. 50-277

1.0 Introduction

The Philadelphia Electric Company (the licensee) has proposed changes to the Technical Specifications of the Peach Bottom Atomic Power Station, Unit No. 2 (Ref. 1). The proposed changes relate to the replacement of 276 fuel assemblies constituting refueling of the reactor core for 6th cycle operation at power levels up to 3293 Mwt (100% power).

2.0 Fuel Design Evaluation

The reload application (Ref. 1) contains five fuel-design-related changes: (1) analysis of the safety considerations involved in the reactor refueling and the Cycle 6 operating limits, (2) continued operation with two previously irradiated fuel assemblies following reconstitution, (3) continued operation with developmental fuel channel boxes, (4) incorporation of new and revised Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the Cycle 6 fuel including extended exposure MAPLHGR limits for standard and lead test assemblies and (5) addition of a generic MAPLHGR curve for General Electric P8X8R fuel.

2.1 Safety Analysis of Cycle 6 Operating Limits

The licensee's analysis of the safety considerations involved in the reactor refueling and the Cycle 6 operating limits are set forth in the Peach Bottom Unit 2 Cycle 6 Reload Report (Ref. 2). In all fuel-design-related areas except those identified in Section 2.0 above, the Reload Report relies on a generic document, the General Electric Reload Fuel Application Report (Ref. 3).

The latter report has been reviewed and approved (Ref. 4) by the NRC staff. We conclude that additional staff review of those portions of Reference 2 concerning the standard fuel design is unnecessary for Cycle 6 operation.

In addition to the 210 standard 8x8R and 552 prepressurized 8x8R fuel assemblies, two previously irradiated high-burnup lead test assemblies (LTAs) will be inserted in the Cycle 6 core. The operation of these lead test assemblies was previously approved through Cycle 5. The safety analysis for operation of these assemblies during Cycle 6 is described in Appendix C of Reference 2.

Although Appendix C generally follows the fuel design criteria used for the standard fuel (Ref. 3), some of the analyses used to demonstrate that the LTAs meet these criteria appear to have been performed at exposures higher than normally encountered. From the description of the LTA design analysis presented in Appendix C, we are unable to determine (a) differences between that analysis and those contained in the General Electric Reload Fuel Application Report (Ref. 3) and (b) differences between that analysis and those identified in the Standard Review Plan (Ref. 5). We note, however, that the design criteria used, and the level of detail presented in Appendix C, are typical of that previously accepted for LTA programs. The analysis is acceptable because (a) the allowable power rating of these assemblies at high exposures is significantly lower than the rest of the core and (b) only two lead test bundles are involved.

2.2 Reconstituted Fuel Assemblies

The licensee has proposed reconstitution and continued use of two previously irradiated standard 8x8R fuel assemblies. The purpose of the reconstitution is to obtain fission gas release data in conjunction with a General Electric extended burnup test program. Six rods in each fuel assembly will be replaced with fresh rods. The twelve removed rods will be subjected to fission gas pressure measurements (puncture tests). An analysis of the safety considerations involved in the continued operation of the reconstituted fuel is described in Appendix B of Reference 2. The licensee has stated that the mechanical design changes in the new rods were minor and that the initial enrichments of the new rods were selected to assure that the power peaking in the reconstituted assemblies will be similar to the non-reconstituted assemblies. Therefore, the results of the fuel rod thermal and mechanical design evaluations in Reference 3 remain applicable to the reconstituted assemblies. We agree with this conclusion and find the use of the reconstituted fuel assemblies for Cycle 6 to be acceptable.

2.3 Developmental Fuel Channel Boxes

The licensee has requested approval for the continued operation of twelve developmental (i.e., experimental) fuel assembly channel boxes, which were initially installed during the first reload of Peach Bottom Unit 2. These channel boxes utilize various wall thicknesses and heat treatments as part of a study of oxide growth and corrosion behavior. The analysis of the safety considerations involved in continuing the use of the developmental channel boxes is presented in Appendix E of Reference 2 and in Reference 6. Neither report specifically identifies any burnup limitation on the analysis although the Cycle 6 Application (Ref. 1) states that a 40 GWd/MtU limit is observed. Based on previous NRC staff approval of this program and on the developmental nature of these channel boxes, we continue to find their use acceptable at Peach Bottom Unit 2.

2.4 MAPLHGR Limits

The licensee has submitted new and revised MAPLHGR limits for all Cycle 6 fuel types including extended exposure limits for standard and LTA fuel. These limits were generated by methods (Ref. 7) submitted as part of the application. Although the methodology used is generally applicable for these limits, we believe that the effects of enhanced fission gas release in high burnup fuel (above 20 GWd/MtU) were not adequately considered in the generic analysis. In response to this concern, the General Electric Company requested (Refs. 8-9) that credit for approved, but unapplied, emergency core cooling system (ECCS) evaluation model changes be used to avoid MAPLHGR penalities at higher burnup.

This proposal was found acceptable (Ref. 10) provided that certain plant-specific conditions were met. In a letter dated July 15, 1981 (Ref. 11), Philadelphia Electric Company found the General Electric proposal applicable to both Peach Bottom Units 2 and 3. On the basis of this finding, we conclude that the MAPLHGR limits proposed for Peach Bottom Unit 2 Cycle 6 are acceptable.

It should be noted that these MAPLHGR limits have been provided for average planar exposures of up to 40 GWd/STU for all fuel types except the two LTAs, which have MAPLHGR limits specified for average planar exposures of up to 50 GWd/STU. We regard these proposed Technical Specifications as limiting on both power and burnup, and therefore the peak planar average exposure during Cycle 6 operation must be limited to 40 GWd/STU for all but the two LTAs by the proposed Technical Specifications. It should also be noted that a basis for MAPLHGR extensions beyond 40 GWd/STU average planar exposure has not yet been accepted by the NRC staff for other than LTA operation. A General Electric submittal that would justify such high-burnup application is expected in the near future. However, communication with the licensee has revealed that average planar exposures beyond 40 GWd/STU are not anticipated for 8DRB284L fuel and, therefore, such a submittal is not required to support the Cycle 6 safety analysis. We thus find the MAPLHGR limits acceptable as submitted.

2.5 Generic MAPLHGR Limit

The licensee has proposed the addition of a generic P8X8R MAPLHGR curve to the plant Technical Specifications. This curve would be added for the purpose of reducing the need for future cycle-dependent revisions to the Technical Specifications. It was constructed to bound the Reload 4 and Reload 5 P8x8R fuel at Peach Bottom Unit 2. Because this curve was generated from a number of specific fuel types, rather than from the set of all possible P8x8R fuel loadings, it is necessary that the licensee determine that the generic P8x8R MAPLHGR curve is bounding for future specific MAPLHGR limits supplied by the fuel vendor (e.g. General Electric). Should this be the case, we would accept use of the generic MAPLHGR curve without additional modification to the plant Technical Specifications.

2.6 Conclusions

We have reviewed those sections of the reload report for Peach Bottom Unit 2, Cycle 6, dealing with changes to the fuel system design and its analysis. We find those portions of the application acceptable.

3.0 Nuclear Design Evaluation

The reload report follows the procedures described in Reference 3. Reference 3 has been approved for use in the nuclear design and analysis of reloads for boiling water reactors (Reference 4). Its use is acceptable for Peach Bottom. Separate cycle-specific analyses were done for the rotated bundle event, the rod withdrawal error and the control rod drop accident. The latter analysis was necessary because the scram curve for Cycle 6 is non-conservative with respect to the generic curve. The results of the cycle-specific analyses meet the relevant criteria and are acceptable.

We have reviewed the nuclear aspects of the fuel assembly rod replacement and extended exposure LTAs. Six fuel rods are to be removed from each of two assemblies and replaced with fresh rods having enrichments that are designed to compensate for the depletion of the removed rods. The licensee concludes that this replacement will have a negligible effect on the nuclear characteristics of the assemblies and of the core. We concur with this conclusion.

Two LTAs, which are part of a program to assess the effect of extended burnup on boiling water reactor fuel, will remain in Peach Bottom in Cycle 6. The effect of the presence of these assemblies on the nuclear characteristics of the core has been analyzed. The reactivity (K_{∞}) of the bundles decreases monotonically with burnup. The local peaking factors tend to increase but the bundle powers decrease (due to the reactivity decrease) so that no limits are approached during the extended burnup. Doppler and void reactivity coefficients remain essentially constant. In summary, the presence of the two LTAs will have a negligible effect on the core nuclear characteristics.

We conclude that the proposed Cycle 6 of the Peach Bottom Unit 2 reactor is acceptable with respect to its core physics aspects.

4.0 Thermal-Hydraulic Evaluation

Peach Bottom Unit 2, Reload 5 fuel assemblies are identical in mechanical design to P8x8R assemblies previously licensed and operated in Peach Bottom-2, Reload 4 (Ref. 12). The new fuel assemblies differ from the existing fuel assemblies in the core only in having higher U-235 enrichment. This change was accounted for in the submitted reload analysis. This review includes the following areas: (1) safety limit Minimum Critical Power Ratio (MCPR), (2) operating limit MCPR, (3) thermal-hydraulic stability, and (4) change to Technical Specifications 3.5.K and 4.5.K.

The objective of this review is to confirm that the thermal-hydraulic design of the reload core has been accomplished using acceptable methods, and provides acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients, and is not susceptible to thermal-hydraulic instability. The thermal-hydraulic models and reload methodology used are described in Reference 3.

4.1 Safety Limit MCPR

The safety limit MCPR has been established to assure that at least 99.9 percent of the fuel rods in the core do not experience boiling transition during the worst anticipated operational occurrence. As stated in Reference 3, the safety limit MCPR is 1.07. There has been no change in the safety limit MCPR for Peach Bottom-2 from Reload 4 to Reload 5.

4.2. Operating Limit MCPR

Various transients could reduce the MCPR below the intended safety limit MCPR during Cycle 6 operation. The anticipated operational transients have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (ΔCPR). The ΔCPR values given in Section 11 (Ref. 2) are plant specific values which include results for the transients calculated by using the ODYN methods (Refs. 13 and 14).

The maximum value of ΔCPR resulting from the limiting transient, the generator load rejection without bypass transient, is 0.36 for Reload 5 as compared to 0.23 for Reload 4 (Refs. 2 and 15). The large difference of Δ CPR for this transient is due to the use of the ODYN methods compared to the REDY methods used in Reload 4.

The calculated ΔCPRs were adjusted to reflect either Option A or Option B ΔCPR by employing the conversion method described in Reference 16. The MCPRs are then determined by adding the Δ CPR to the safety limit.

Section 11 of Reference 2 presents the Δ CPR for both non-pressurization and pressurization events. The maximum MCPRs calculated by using the ΔCPR values in Section 11 are specified as the operating limit MCPRs and are incorporated in the Technical Specifications. We have reviewed the operating limit MCPR results discussed above and found these results acceptable.

4.3 Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (Ref. 2) show that the maximum thermal-hydraulic stability decay ratio is 0.85 for Reloads 5 and 4. Because operation in the natural circulation mode will be prohibited by Technical Specification 2.1.A, there will be added margin to the core stability and this is acceptable to the NRC staff.

4.4 Changes to Technical Specifications 3.5.K and 4.5.K

The operating limit MCPR Technical Specification has been modified to include an Option B format where the operating limit MCPRs vary with the measured scram time (τ). The specification is based on measurements to the 20 percent inserted position. Figures 3.5.K.1, 3.5.K.2 and 3.5.K.3 of the proposed Technical Specifications show operating limit MCPR vs τ for 8x8 LTA, P8x8R and P8DRB285 fuels, respectively.

We find that the approved ODYN methods (Refs. 13 and 14) were used and that the results of analyses are consistent with the proposed operating limit MCPR to avoid violation of the safety limit MCPR for the design transients. We conclude that this core reload will not adversely affect the capability for safe operation during Cycle 6 and that the proposed changes to Technical Specifications 3.5.K and 4.5.K. discussed above are acceptable.

5.0 Cycle 6 Transient Analyses

Generic information relative to the reload analyses of boiling water reactor fuel is presented in General Electric Licensing Topical Report NEDE-24011-P-A, "Generic Reload Fuel Applications," July 1979 (Ref. 3). This report is supplemented by plant-specific information contained in References 2 and 7. Together these documents provide the bases for the licensee's safety analysis for Reload 5 and the proposed Technical Specification changes associated with the reload (Cycle 6).

The licensee stated (Ref. 2) that all transients that are the basis of the Peach Bottom-2 Final Safety Analysis Report were reviewed for Cycle 6 and that those transients that were critical with respect to safety margins and sensitive to the core related parameter changes were reanalyzed.

The ODYN code is used to define input parameters for Critical Power Ratio (CPR) calculations during rapid pressurization transients. The ODYN code also is used to calculate the pressure transients more accurately than the REDY code and provides more detailed outputs. In Reference 2, the licensee has provided graphical results from the analysis of pressurization transients.

The transients included are:

(1) Main Steam Isolation Valve Closure

(2) Feedwater Controller Failure

(3) Generator Load Rejection Without Bypass

The licensee has performed the required analyses. As a consequence of using more than one code (ODYN and REDY) for the transients and the two options available with ODYN (Option A with straight penalty for uncertainties and Option B with statistical convolution of uncertainties and rod scram times), the limiting transient for Peach Bottom-2 is dependent upon periodic on-site measurements of average scram time. Depending upon the measured average scram time, the MCPR operating limit will change as shown on TS Figures 3.5.K.1, 2 and 3.

MCPRs are adjusted using Option B when all scram specifications in section 4.5.K.2.a. of the plant Technical Specifications are met. For operating limit MCPR values, see TS Table 3.5.K.2. In the event that the scram time specification is not met, a linear interpolation between the Option A MCPR and the Option B MCPR will be performed as given in Ref. 16. Operating limit MCPRs adjusted using Option A are given in TS Table 3.5.K.3.

We have reviewed the General Electric generic scram time specification procedure using ODYN (Options A and B) and have found it to be acceptable (Ref. 17 and 18). The licensee has duplicated these procedures, and we conclude that this is acceptable. We have also reviewed the licensee's proposed changes to Technical Specifications related to MCPR (pages 133b, 133c, 133d, 133e, 142, 142a and 142b) and conclude that these changes are acceptable.

6.0 Incorporation of the 67B Control Rod Scram Time

This Technical Specification change proposes that the 67B Control Rod Drive (CRD) scram times be incorporated into the Technical Specifications.

The 67B CRD scram times, in replacing the 67A scram times, require a 3.5 second average scram insertion time, rather than 5.0 second average scram insertion time for the 90% inserted, from the fully withdrawn position.

This change is proposed to be in conformance with the reload-unique transient analysis input utilized in Reference 2. The change is acceptable because the results of the transient analysis incorporating the change meet acceptable criteria for operating limit MCPRs as discussed above.

7.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

8.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 17, 1982

The following NRC personnel have contributed to this Safety Evaluation:
Morton Fairtile, George Thomas, Summer Sun, John Voglewede and Walter Brooks.

REFERENCES

1. E. J. Bradley (PECO) letters to H. R. Denton and J. F. Stolz (NRC) dated February 19, 1982 and June 3, 1982.

2. "Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit 2, Reload No. 5," General Electric Company Report Y1003J01A34, December 1981.

3. "General Electric Boiling Water Reactor Generic Reload Fuel Applications," General Electric Company Report NEDE-24011-P-A-2 (Proprietary) and NEDO-24011-A-2 (Non-Proprietary), July 1981.

4. D. G. Eisenhut (NRC) letter to R. Gridley (GE) dated May 12, 1978.

5. U.S. Nuclear Regulatory Commission Standard Review Plan Section 4.2 (Rev. 2), "Fuel System Design," U.S. Nuclear Regulatory Commission Report NUREG-0800 (formerly NUREG-75/087), July 1981.

6. "Developmental Channels-Supplemental Information for Reload 1 Licensing Submittal for Peach Bottom Atomic Power Station Unit 2," General Electric Company Report NEDO-21172, Rev. 1, Supplement 2, March 1976.

7. "Loss-of-Coolant Accident for Peach Bottom Atomic Power Station Unit 2," General Electric Company Report NEDO-24081, December 1977 (with Addenda 1-7).

8. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 6, 1981.

9. R. E. Engel (GE) letter to T. A. Ippolito (NRC) dated May 28, 1981.

10. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "Extension of General Electric Emergency Core Cooling Systems Performance Limits" dated June 25, 1981.

11. S. L. Daltroff (PECo) letter to J. F. Stolz (NRC) dated July 15, 1981.

12. Letter from R. Reid (NRC) to E. Bauer (PECo) dated June 13, 1980.

13. Letter from J. Quirk (GE) to P. Check (NRC), ODYN Improvements, September 25, 1981.

14. Letter from J. Quirk (GE) to T. Speis (NRC), ODYN Improvements, October 13, 1981.

15. Supplemental Reload Licensing Submittal for Peach Bottom Atomic Power Station Unit No. 2, Reload No. 4, NEDO-24237-A, dated February 1980.

16. Letter from R. Buchholz (GE) to P. Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits," dated January 19, 1981.

17. "Safety Evaluation for the General Electric Topical Report Qualification on the One-Dimensional Core Transient Model for Boiling Water Reactors," NEDO-24154 and NEDE-24154P Volumes I, II, and III, June 1980.

18. "Supplemental Safety Evaluation for the General Electric Topical Report Qualification of the One-Dimensional Core Model for Boiling Water Reactors," NEDO-24154 and NEDE-24154P Volumes I, II, and III, January 1981.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-277

PHILADELPHIA ELECTRIC COMPANY, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 86 to Facility Operating License No. DPR-44, issued to Philadelphia Electric Company, Public Service Electric and Gas Company, Delmarva Power and Light Company, and Atlantic City Electric Company, which revised Technical Specifications (TSs) for operation of the Peach Bottom Atomic Power Station, Unit No. 2 (the facility) located in York County, Pennsylvania. The amendment is effective as of the date of issuance.

The amendment changes the TSs to permit Cycle 6 operation of the facility.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated February 19, 1982, as supplemented June 3, 1982, (2) Amendment No. 86 to License No. DPR-44 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, DC and at the Government Publications Section, State Library of Pennsylvania, Education Building, Commonwealth and Walnut Streets, Harrisburg, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 17th day of June 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing